Duquesne Light Company Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004

RONALD L. LeGRAND
Division Vice President -
Nuclear Operations and Plant Manager

(412) 393-7622
Fax (412) 393-4905

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L-97-022

Beaver Valley Power Station, Unit No. 1
Docket No. 50-334 License No. DPR-66
LER 97-025-00

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

In accordance with Appendix A, Beaver Valley Technical Specifications, the following Licensee Event Report is submitted:

LER 97-025-00, 10 CFR 50.73(a)(2)(iv), "Ground in Feedwater Flow Controller Results in High Steam Generator Level and Subsequent Turbine Trip/Reactor Trip."

Ronald L. LeGrand

R. L. LeGrand

Attachment

IE22

cc: Mr. H. J. Miller, Regional Administrator
United States Nuclear Regulatory Commission
Region 1
475 Allendale Road
King of Prussia, PA 19406

Mr. D. S. Brinkman
BVPS Licensing Project Manager
United States Nuclear Regulatory Commission
Washington, DC 20555

Mr. David Kern
BVPS Senior Resident Inspector
United States Nuclear Regulatory Commission

Mr. J. A. Hultz
Ohio Edison Company
76 S. Main Street
Akron, OH 44308

Mr. Steven Dumek
Centerior Energy Corporation
6670 Beta Drive
Mayfield Valley, OH 44143

INPO Records Center
700 Galleria Parkway
Atlanta, GA 30339-5957

Mr. Michael P. Murphy
Bureau of Radiation Protection
Department of Environmental Protection
RCSOB-13th Floor
P.O. Box 8469
Harrisburg, PA 17105-8469

Director, Safety Evaluation & Control
Virginia Electric & Power Company
5000 Dominion Blvd.
Innsbrook Tech. Center
Glen Allen, VA 23060

CATEGORY 1

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FACIL:50-334 Beaver Valley Power Station, Unit 1, Duquesne Light C 05000334
AUTH.NAME AUTHOR AFFILIATION
HART,R.D. Duquesne Light Co.
LEGRAND,R.L. Duquesne Light Co.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: LER 97-025-00:on 970807,ground in feedwater flow controller results in high SG level & subsequent turbine trip/reactor trip was discovered.Caused by manufacturing defect on circuit card.Event response team formed.W/970904 ltr.

DISTRIBUTION CODE: IE22T COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 7
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NRC FORM 366 (4-95) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

APPROVED BY OMB NO. 3150-0104
EXPIRES 04/30/98
ESTIMATED BURDEN PER RESPONSE TO COMPLY WITH THIS INFORMATION COLLECTION REQUEST: 50.0 HRS. FORWARD COMMENTS REGARDING BURDEN ESTIMATE TO THE INFORMATION AND RECORDS MANAGEMENT BRANCH (MNBB 7714), U.S. NUCLEAR REGULATORY COMMISSION, WASHINGTON, DC 20555-0001, AND TO THE PAPERWORK REDUCTION PROJECT (3150-0104), OFFICE OF MANAGEMENT AND BUDGET, WASHINGTON, DC 20503.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Beaver Valley Power Station Unit 1	05000334	1 OF 5

TITLE
Ground in Feedwater Flow Controller Results in High Steam Generator Level and Subsequent Turbine Trip/Reactor Trip

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
08	07	97	97	025	00	09	04	97	N/A	

OPERATING MODE (9)	1	20.402(b)	20.405(c)	X 50.73(a)(2)(iv)	73.71(b)
POWER LEVEL (10)	100%	20.405(a)(1)(i)	50.36(c)(1)	50.73(a)(2)(v)	73.71(c)
		20.405(a)(1)(ii)	50.36(c)(2)	50.73(a)(2)(vii)	OTHER
		20.405(a)(1)(iii)	50.73(a)(2)(i)	50.73(a)(2)(viii)(A)	(Specify in abstract below and in Text NRC Form 366A)
		20.405(a)(1)(iv)	50.73(a)(2)(ii)	50.73(a)(2)(viii)(B)	
		20.405(a)(1)(v)	50.73(a)(2)(iii)	50.73(a)(2)(x)	

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (include Area Code)
R. D. Hart, Senior Licensing Supervisor	(412) 393-5284

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS		CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO NPRDS
B	JB	FFC	W120	Y						

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
YES (if yes, complete EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limited to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On August 7, 1997, at 2018 hours, Beaver Valley Power Station Unit 1 was in Mode 1 at 100% power when the unit experienced an automatic turbine trip, reactor trip and feedwater isolation (FWI). The feedwater regulating valve for "A" steam generator (S/G) failed open during the replacement of a resistor block in the S/G level control circuit by Instrumentation and Controls (I&C) technicians. The spurious feedwater regulating valve opening resulted in a rapid rise in S/G level. When Control Room Operators attempted to take manual control of the valve, it would not respond.

When 2 out of 3 level channels on the "A" S/G reached the Hi-Hi setpoint, a full FWI was generated, which also tripped the turbine. The FWI secured both main feedwater pumps and closed the feedwater isolation valves. The turbine trip initiated a reactor trip at 2018 hours. The Control Room crew responded by implementing the required emergency operating procedure (EOP) for reactor trip. S/G level shrinkage from the secondary pressure rise resulted in the expected automatic start of all three auxiliary feedwater pumps. Approximately 30 seconds after the turbine trip, the main generator tripped and a fast bus transfer to the system transformers occurred. The emergency diesel generators started automatically due to a transient bus undervoltage condition, but load shedding and sequencing of loads onto the emergency buses was not required. The reactor trip EOP was exited approximately 3 minutes after the trip and the unit was stabilized and placed in a safe, controlled shutdown condition with no safety implications to the health and safety of the public.

At 2220 on August 7, 1997, a four hour non-emergency notification was made pursuant to the requirements of 10CFR50.72(b)(2)(ii). This report is being made pursuant to the requirements of 10CFR50.73(a)(2)(iv).

An Event Response Team was formed on August 8, 1997 to investigate the reactor trip. The ERT determined that I&C and Operator actions were correct and did not cause the trip. The root cause of the trip was determined to be the existence of excess solder on a circuit card in the S/G feedwater flow control module. The excess solder came in contact with the module chassis, creating an internal ground on the feedwater control loop. Ground current initiated a full open demand signal to the "A" main feedwater regulating valve and also prevented operators from taking manual control of the valve. A new module was installed to replace existing S/G feedwater flow control module FC-FW-478 and was demonstrated to function properly via post maintenance testing on August 9, 1997.

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TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

PLANT AND SYSTEM IDENTIFICATION

Westinghouse Pressurized Water Reactor (PWR)

Main Feedwater Steam Generator Level Control System {JB}

Main Feedwater Regulating (Flow Control) Valve FCV-FW-478 {JB/FCV}

Steam Generator Flow Controller FC-FW-478 Westinghouse (Hagen) Model 124 {JB/FFC}

Current to Voltage Converter Input Module LC-478 B/R I/V {JB/CNV}

* Energy Industry Identification System (EIIS) codes and component function identifier codes appear in the text as {SS/CCC}

DESCRIPTION OF THE EVENT

On August 7, 1997, at 2018 hours, Beaver Valley Power Station Unit 1 was in Mode 1 at 100% power when the unit experienced an automatic turbine trip, reactor trip and feedwater isolation (FWI). On dayshift prior to the trip, several "A" steam generator level (S/G) deviation alarms had been received. Operations informed Instrumentation and Controls (I&C) at approximately 0730 hours that the "A" S/G level was controlling at 49% versus the required 44%.

During the afternoon shift, the deviation alarms became more frequent and longer in duration. I&C investigations identified an apparent discrepancy in readings on current to voltage converter module LC-478 B/R I/V {JB/CNV} in the S/G level control system {JB}, in comparison to other I/V modules, and initially concluded that, either some leads may have been loose, or internal resistors degraded. Operators placed main feedwater regulating valve FCV-FW-478 {JB/FCV} for the "A" S/G in manual control, while I&C tightened the connections on the resistor block on the current to voltage converter in the level control circuitry, which was believed to be the source of the level deviations. This did not correct the anomaly.

I&C then proceeded to change the resistor block. During this evolution, the "A" main feedwater regulating valve failed open. The spurious feedwater regulating valve opening resulted in a rapid rise in S/G level. Operators attempted to take manual control of the feedwater regulating valve, but it would not respond. When 2 out of 3 level channels on the "A" S/G reached the Hi-Hi setpoint, a full FWI was generated, which tripped the turbine, secured both main feedwater pumps and closed the feedwater isolation valves. The turbine trip above Permissive P-9 tripped the reactor at 2018 hours.

The Control Room crew entered Emergency Operating Procedure (EOP) E-0, "Reactor Trip or Safety Injection." S/G level shrinkage from the secondary pressure rise resulted in the expected automatic start of all three auxiliary feedwater pumps. The AMSAC system properly operated to provide a backup turbine trip and AFW pump start signals.

Approximately 30 seconds after the turbine trip, the main generator tripped and a fast bus transfer to the system transformers occurred. The emergency diesel generators (EDGs) started automatically due to a transient bus undervoltage condition, but load shedding and sequencing of loads onto the emergency buses was not required. Normal bus transients during reactor trip may cause EDG starts, based upon the existing relay setpoints.

Downscale indication of the level pen on the "A" S/G level recorder and a continuous open signal to the feedwater regulating valve (with control in manual and zero demand input) remained after the trip. EOP E-0 was exited, approximately 3 minutes after the trip, and the unit was stabilized and placed in a safe, controlled, shutdown condition with no safety implications to the health and safety of the public.

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CAUSE OF THE EVENT

The root cause of the trip was determined to be a manufacturing defect on a circuit card, i.e., the existence of excess solder on the limiter circuit card within S/G feedwater flow controller module FC-FW-478. The excess solder came in contact with the module chassis bracket, creating an internal ground on the feedwater control loop. Ground current initiated a full open demand signal to the "A" main feedwater regulating valve FCV-FW-478 and defeated the manual signal, which prevented operators from taking manual control of the valve. The resultant excessive feedwater flow caused the "A" S/G level to rise rapidly, which resulted in a coincident two out of three S/G level Hi-Hi condition, which caused a turbine trip and FWI. The turbine trip occurring at a power level greater than the P-9 interlock setpoint caused a reactor trip. Downscale indication of the level pen on the "A" S/G level recorder and a continuous open signal to the flow control valve (with control in manual and zero demand input) was also caused by the solder ground. ERT investigations determined that the trip was not caused by Operator actions or I&C troubleshooting and maintenance activities.

ANALYSIS OF THE EVENT

Following the trip, an Event Response Team (ERT) was formed to evaluate the reactor trip. As a part of the ERT evaluation, an investigation was performed to evaluate the response of the feedwater control system.

As documented in the maintenance work request (MWR) implemented by I&C to investigate the "A" S/G level deviations, current measurements were taken throughout the loop. These measurements indicated the presence of a ground loop within the control loop. During the investigation, individual modules were systematically disconnected from the loop in an attempt to isolate the grounded component. This resulted in the identification of two grounds. The first was the installed ground. The second was located internally to the flow controller FC-FW-478 for feedwater regulating valve FCV-FW-478. This module had been replaced on August 1, 1997. The flow controller was removed from service and inspected on the bench.

Testing revealed a soft ground between two pins on the connector on the rear of the module. As the module itself, as well as internal wiring, was moved, the value of the resistance reading to ground changed. The ground was localized to a limiter board in the controller module. When the board was removed and visually inspected, it was discovered that an excessive amount of solder on one solder joint was contacting the chassis of the controller. I&C demonstrated that with this ground present, the output of controller FC-FW-478 fails to a value which can cause "A" S/G feedwater regulating valve FCV-FW-478 to go to the full open position. This will occur regardless of whether resistor block LC-478 B/R is wired in the loop. Also, the raise/lower buttons of the auto/manual station will no longer function, resulting in loss of control of the feedwater regulating valve.

Investigations by Westinghouse and Duquesne Light Company (DLCo) indicated that the ground probably existed from initial manufacture of the module. It was concluded that, as the module operated in the loop over the several days that it was installed, the value of the ground slowly hardened (decreased in resistance). This is evidenced by the increased S/G level control point that was observed by Operations and the fact that the deviation continued to increase over the course of the day on August 7, 1997. The investigations concluded, after extensive review of the schematics for the controller and the loop, that the events surrounding the trip, and the equipment responses following the trip can be explained by the ground in the flow controller. It was also concluded that lifting the leads to resistor block LC-478 B/R in the level control circuitry had no impact on the ground. I&C and Operator actions were determined to be correct and did not cause the trip.

The autostart of the Emergency Diesel Generators was determined to be the result of the transient bus undervoltage condition. Normal bus transients during reactor trip may cause EDG starts, based upon the existing relay setpoints and timing.

The ERT performed an evaluation of secondary plant-related anomalies observed during the post trip recovery and concluded that these anomalies did not contribute to the cause of the trip, nor did they significantly impact post trip recovery or plant stabilization efforts. These issues are being addressed under the plant Corrective Action Program.

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CORRECTIVE ACTIONS

1. An Event Response Team (ERT) was formed on August 8, 1997 to review and evaluate the trip. The ERT completed the evaluation and documented its findings in a report on August 9, 1997.

2. DLCo I&C conducted an investigation into the feedwater flow anomaly and identified the root cause to be a ground caused by excess solder in feedwater flow control module FC-FW-478 on August 8, 1997.

3. A new module was installed to replace S/G feedwater flow control module FC-FW-478 and was demonstrated to function properly via post maintenance testing on August 9, 1997.

4. An extent of condition review was completed by August 22, 1997 to identify all plant applications of Model 124 Westinghouse (Hagen) flow and pressure controllers using the same part number and assess the potential impact of a similar failure. These were determined to exist in multiple non-safety applications in other systems. The ERT evaluation determined that this was an isolated manufacturing defect unique to the FC-FW-478 controller installed in the BVPS Unit 1 feedwater control system. There are no applications of this type of controller at Unit 2.

FOLLOW-UP CORRECTIVE ACTIONS

1. A screening evaluation of the defective feedwater control module issue is being performed pursuant to the requirements of 10CFR21 and is being tracked under the Corrective Action Program. This evaluation will be completed by September 12, 1997. If determined reportable, a Part 21 assessment will be provided in a supplement to this LER.

REPORTABILITY

At 2220 on August 7, 1997, a four hour non-emergency notification was made pursuant to the requirements of 10CFR50.72(b)(2)(ii). This report is being made pursuant to the requirements of 10CFR50.73(a)(2)(iv) as "Any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)..."

SAFETY IMPLICATIONS

In response to the trip, the appropriate plant emergency procedures were entered and the unit placed in a safe, controlled shutdown condition. Hence there were no safety implications to the health and safety of the public as a result of this event.

SIMILAR EVENTS

A review of Licensee Event Reports for the past two years identified the following similar events:

1. LER 2-97-001, "Reactor Trip Due to Main Transformer Ground Protection Relay," dated February 3, 1997.

2. LER 1-97-005, "Inadvertent Operation of 345 KV Bus Backup Timer Relay Results in Dual Unit Reactor Trips," dated April 14, 1997.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

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ADDITIONAL INFORMATION

Unit 1 Emergency Diesel Generator Performance Data

Unit Data

Sum of start and load run failures for the last 20 start/load demands 0

Sum of start and load run failures for the last 50 start/load demands 1

Sum of start and load run failures for the last 100 start/load demands 2

Individual Emergency Diesel Generator Data

	EDG EE-EG-1	EDG EE-EG-2
Failures in the last 25 start demands	1	0
Failures in the last 25 load demands	0	0